Docket

JAN 4 1974

Docket Nos. 50-438
and 50-439

Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

In order that we may continue our review of your application for a license to construct the Bellefonte Nuclear Plant, Units 1 and 2, we are providing Regulatory staff positions regarding pertinent safety matters. These positions are listed in Enclosure No. 1. We request that you state your intent regarding compliance with each of these positions and amend your application accordingly. We are prepared to meet with you to facilitate a complete understanding of these safety matters and the bases for our positions.

We have requested in Enclosure No. 2, additional information needed to clarify and amplify previously submitted information.

In order to maintain our licensing review schedule for those matters dealt with in the enclosures, we will need completely adequate responses by February 14, 1974, to all of these staff positions and requests for information.

Please inform us within 7 days after receipt of this letter of your confirmation of the schedule date or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our request, it is highly likely that the overall schedule for completing the licensing review for the project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to returning to this project, the extension will most likely be greater than the delay in your response.

Please contact us if you have any questions regarding the staff positions or the information requested.

Sincerely,

/s/

A. Schwencer, Chief
Light Water Reactors Branch 2-3
Directorate of Licensing

LB

Enclosure and ccs: See next page

OFFICE						
SURNAME						
DATE						

Enclosures:
1. Staff Positions
2. Request for Additional Information

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ENCLOSURE NO. 1
POSITIONS REGARDING CONSTRUCTION PERMIT
TENNESSEE VALLEY AUTHORITY
BELLEFONTE NUCLEAR PLANT, UNITS 1 AND 2
DOCKET NOS. 50-438 AND 50-439

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.69 Your response to Request 3.26 is not adequate. It is the Staff's position that either the finite element method or multiple soil spring approach should be used for the soil-structure interaction analysis associated with Category I structures that are supported by a relatively thin layer of soil either with or without shallow embedment. It is noted that based upon the theory of elastic half space, the rather thin soil layer and the use of the reinforced concrete retaining wall structure as depicted by Figure 3.8-40C for the Category I borated water storage tank structure will invalidate the adoption of the single soil spring approach. State your intent to adopt the above mentioned position and indicate the date at which the preliminary results of this analysis will be made available.

3.70 Your response to Request 3.30 is not adequate. Figure 3.7-22 shows a suspended slab between elevations 580.0 and 610.0. This suspended slab (El.590.0±) implies some flexibility in the vertical direction. This is inconsistent with your assumption of a fixed point set at the El. 610.0 slab. It is the Staff's position that you should reflect the above mentioned vertical flexibility in the dynamic modeling of the auxiliary control building.

3.71 Delaying your response to request 3.37 until the FSAR is not acceptable. Provide a response to this request.

ruary 14, 19

3.72 The last sentence of Section 3.7.2.1.1 (page 3.7-2a) and the first sentence of the last paragraph on page 3.7-3 should be revised to reflect three dimensional modeling of Category I seismic system analysis, so that the Staff's position on the three component seismic response combination procedure indicated in your revised page 3.7-3 can be effectively implemented.

3.73 The response to Request 3.65 is not adequate since you present no basis for your response. It is the Staff's position that multi-element seismoscopes which can promptly provide the response spectra to the plant operator are required. These spectra can then be compared with the site design response spectra to aid the operator in taking appropriate action if an earthquake larger than a prescribed level were to occur. These instruments can also aid in verifying the analytical models and damping factors used in the seismic analysis. A new Regulatory Guide on seismic

instrumentation based upon a recent paper* is being prepared. State your intent to provide such an instrumentation system. Also specify number, type and location of the instrumentation and the basis for selecting specific instrument locations.

3.74 The response to Request 3.40 is not adequate since sufficient justification has not been provided for the proposed deviation from Regulatory Guide 1.35. As indicated in the Guide, containments that differ from the defined "typical" prestressed concrete containments should have a tendon surveillance program which follows the Guide as the basis for a comparable program. While the opportunity exists to provide justification for other programs, it is the opinion of the Regulatory staff that insufficient historical evidence or experience is available to warrant any significant change in the surveillance program outlined in Regulatory Guide 1.35. Provide additional information to support the contention that a reduced program of tendon surveillance is adequate or adopt a program similar to that in Regulatory Guide 1.35.

3.75 The response to Request 3.46 is not adequate. The statement in Section 3.7.2.1.1.3 that the turbine and service buildings will be designed in a manner similar to the requirements of the Uniform Building Code does not present sufficient justification to guarantee that these structures will not collapse and possibly damage adjacent Category I structures. Provide sufficient information to demonstrate that:

(1) the collapse of any non-Category I structure will not strike a seismic Category I structure or component,

(2) the collapse of the non-Category I structure(s) will not impair the integrity of Seismic Category I structures or components, or

(3) the non-Category I structures will be designed to prevent their failure under seismic conditions in a manner such that the margin of safety of these structures is equivalent to that of Category I structures.

3.76 The responses to Requests 3.44 and 3.49 are not adequate. The margin of safety utilized in the structural design criteria for seismic Category I structures inside and outside the containment as presented in the PSAR is not equivalent to that presented in Enclosure 2, "Structural Design Criteria for Category I Structures Outside the Containment" to the Staff's September 14, 1973 request for information. It is the opinion of the Staff that Equations 2, 2a, 3 and 3a as stated on Pages 3.8-55, 56 and 64a, Equation 2 as stated on Page 56A and Equations 2 and 3 as stated

*Kapur, K. K., "Seismic Instrumentation for Nuclear Power Plants", Paper presented at the ANS Water Reactor Safety Conference, March 26-28, 1973.

on Page 65 are not adequately conservative. As indicated in Enclosure 2 of our September 14, 1973 letter, it is the Staff's position that the 33% increase in allowable stresses for concrete and steel due to seismic or wind loadings should not be permitted for service load conditions. This position is also reflected in the load factors for the Strength Design method. Provide additional information to support the contention that your set of criteria is adequate or adopt the staff's criteria.

13.0 CONDUCT OF OPERATIONS

13.7 You state in Section 13.1.3.1 and show in Figs. 13.1-4 and 13.1-5 that under certain conditions, the Power Plant Operations Supervisor may not need to hold a senior operator license. It is the Staff's position that the Power Plant Operations Supervisor's position is comparable to the position of Operations Manager described in ANSI N18.1 and therefore should hold a senior operator's license.

13.8 You state in Section 13.1.3.1 that, at the time of initial fuel loading, your assistant unit operators will, as a minimum, have had several weeks onsite plant familiarization. Our position is that at least two assistant unit operators should have at least several months onsite plant familiarization.

ENCLOSURE NO. 2
REQUEST FOR ADDITIONAL INFORMATION
TENNESSEE VALLEY AUTHORITY
BELLEFONTE NUCLEAR PLANT, UNITS 1 AND 2
DOCKET NOS. 50-438 AND 50-439

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.77 The response to Request 3.54 requires clarification in these areas:

a. Section 3.6.1 addresses "Piping components not designed to Code requirements defined in the report." Clarify whether the referenced report is the deleted TDA-TR-1 and whether there are any such piping components not designed to code requirements.

b. Section 3.6.2 is not necessarily in agreement with Regulatory Guide 1.46 in that the method used to determine the type of break has not been brought into accord. Clarify and justify any remaining differences.

3.78 The response to Request 3.55 is not adequate. Expansion or modification is required in the following areas:

a. Identify the piping components whose design is based on actual test if the calculated stresses exceed the allowable limits set by ASME Code as stated in the Amendment 5, first paragraph of Section 3.6.3. Provide a commitment to summarized in the FSAR the comparison between analytical results and test results for strains, stresses, deflections, etc. if these tests are to be performed. Include in this summary a re-examination of the assumptions made in the analyses and justification of significant differences between the analytical and test results.

b. Describe the typical mathematical models used for the dynamic analysis of the piping and associated supports (broken loop).

c. Either modify the equation on page 3.6-2 to adequately characterize the jet thrust during the transient phase prior to reaching steady state flow or provide justification for this omission.

d. The use of a jet flow model employing two zone divergent flow is not sufficiently conservative. An acceptable model employs one zone divergent flow with the half-angle dispersion of 10 degrees.

e. An increase of 20% of the yield strength of the material is not acceptable. A 10% increase of minimum specified design yield strength (Sy) may be used in the analysis to account for strain rate effects. (Refer to Enclosure 3 of the Staff's September 14, 1973 request for information.)

3.79 The response to Request 3.56 with respect to Section 3.6.4 is not adequate. Describe the dynamic analysis methods and procedures which will be used to establish the design adequacy of pipe whip restraints and to determine the loadings imparted by the unrestrained whipping pipe to impacted structures.

Acceptable analytical procedures have previously been forwarded as Enclosure 3 to the Staff's September 14, 1973 request for information. State your intention to conform to Enclosure 3 or justify adequacy of the methods and procedures used. Provide a commitment to discuss the results of these methods in the FSAR.

3.80 The response to Request 3.58 requires amplification in the following area:

In Section 3.9.1.1 describe the different flow modes of operation and transients to which the components and piping systems will be subjected during the preoperational test program. Provide a commitment to discuss the results of these tests in the FSAR. An acceptable response may be developed by referencing similar tests performed on a prototype plant and comparing flow characteristics at different ratings and other significant parameters causing flow-induced vibrations between both plants. Include the dynamic effects which result from the turbine stop valve closure and relief valve operation on the reactor coolant loop. Regulatory Guide 1.68 provides additional guidance.

3.81 The response to Request 3.59 with regard to Sections 3.7.2.1 and 3.9.1.2 is not adequate. Stress limit criteria as described in Section 3.7.2.1 which may assure the pressure-retaining or structural integrity of fluid containing components in the operating mode does not necessarily prevent deformations of sufficient magnitude to interfere with the operability of components and equipment. Demonstrations of operability must in many instances be accomplished by testing in addition to stress and deformation analyses.

Enclosure 4 to the staff's September 14, 1973 request for information outlines an acceptable testing program for seismic qualification of electrical and mechanical equipment. State your intent to employ a suitable testing program to guarantee the operability of active mechanical components under faulted plant condition loads. Identify and justify any differences between your program and that described in the enclosure. If stress and deformation analyses without testing are used, justify the qualification criteria.

3.82 The response to Request 3.60 requires amplification in the following areas:

a. A prototype plant should be designated as soon as it is practicable. Define this prototype or the schedule for its definitive.

b. Amplify your discussion of the method for confirming the validity of the mathematical model used in Sections 3.9.1.3 and 3.9.1.4 for the dynamic analysis of the reactor internals under LOCA loading by comparison (i.e. frequencies and mode shapes) with the results of the reactor internal preoperational vibration testing programs. Discuss the sensitivity of the results to input parameters and explain the methods used for evaluating these key input parameters.

c. Provide the names, descriptions and validation of the computer programs which will be used for the LOCA dynamic analysis. The validation of the computer programs is outlined in Enclosure 10 to the Staff's September 14, 1973 request for information.

d. A dynamic system analysis for confirming the structural design adequacy of the reactor internals (including fuel element assemblies, control rod assemblies and drives) and reactor coolant piping (unbroken loop) to withstand dynamic effects under a simultaneous occurrence of LOCA and SSE is required. Provide a commitment to perform this analysis and a summary of the acceptance criteria to be employed.

e. Either designate an acceptable prototype plant or supplement Section 3.9.1.3 of the PSAR by providing a more detailed description of the preoperational vibration test program which will be used to confirm the design adequacy of the reactor internals under operational transients. Discuss the conformance of the test program to the positions of Regulatory Guide 1.20. The program should include (a) identification of flow transients used for the test, (b) sensor types, number, and locations which will be used to monitor flow excitations and responses of reactor internals, (c) procedures for subsequent visual inspections, and (d) the scheme of cold and hot flow test. If a prototype plant is designated provide a detailed confirmatory test program for Bellefonte Units 1 & 2. If the elements of the test program differ substantially from the requirements of Regulatory Guide 1.20, provide the basis for these differences.

f. In Section 3.9.1.5, describe the forcing functions that will be used for the LOCA dynamic analysis including pressure differentials, direction, rise time, magnitude, duration and initial conditions. Summarize the results of the dynamic analysis including loading combinations which govern the design of the system.

3.83 The response to Request 3.61 is not adequate. State your intention to meet the criteria outlined in Regulatory Guide 1.48 or provide justification for differences between the proposed criteria and the criteria of Regulatory Guide 1.48.

3.84 The response to Request 3.62 is not adequate. State your intention with regard to demonstrating operability of ASME Class 2 and 3 prototype pumps and valves by full-scale full-assembly testing under combined loading conditions in accord with enclosures 6 and 7 previously forwarded with the staff's September 14, 1973 request for information. As an alternate, acceptable criteria may be found in the response to Position 10 for the Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413/414.

3.85 The response to Request 3.66 is not adequate. State the specified maximum allowable support deformation limit that will be used for the most adverse loading conditions.

3.86 The response to Request 3.64 is not adequate. The response refers to a new topical report similar to BAW-10003 that will consider a previously-submitted attachment and the topical report evaluation by the AEC. Provide the submittal date for this topical report.

3.87 Your response to Request 3.36 requires additional information to address stresses in buried piping systems due to free field seismic wave propagation. Describe the procedures for obtaining axial, shear and bending stresses due to differential movements at anchor points of buried piping systems (with or without bends). Discuss the effects of soils surrounding a buried piping system in your dynamic and seismic analysis. Provide the reference for the stiffness expression (page 3.7-16) that was included in your response.

4.0 REACTOR

4.8 The response to Request 4.5 is not adequate, Additional information is required in the following areas:

a. The behavior of a control element within the "split" or "c" guide tubes while being driven against an obstruction has not been addressed. Discuss any potential for "lockup" of the control element into the guide tube slot(s) when forced into a restrained buckling mode by the postulated obstruction,

b. List the operational history of the control system design.

4.9 The computer code HYTRAN is used in the analysis of flow instability in a reactor flow channel. The description provided in the response to Request 4.7 is not of sufficient detail to permit evaluation of the accuracy and applicability of the code. Provide a commitment to supply a topical report describing the mechanics and the experimental verification of the code before application for an operating license.

4.10 Explain the coupling between the SPLIT program (see Request 5.15) and the HYTRAN program. In addition, provide a discussion that may be qualitative in nature that identifies those characteristics of the Bellefonte core design that should have a tendency to reduce the potential for hot channel flow instability.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.18 The response to Request 5.12 is not adequate. State your intention to adopt an acceptable operability assurance program for ASME Class 1 active valves and define the program. Acceptable criteria may be found in the response to Position 10 for the Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413/414 and in enclosures 8 and 9 to the Staff's September 14, 1973 request for information.

6.0 ENGINEERED SAFETY FEATURES

6.78 The response to Request 6.26 is inadequate. It is necessary to analyze small break loss of coolant accidents, provide or reference such analyses.

6.79 The response to Request 6.27 is inadequate. In regard to operator actions required to mitigate the consequences of accidents, our position is that operator actions should be reduced where practicable and ECCS component reliability kept at a high level. The normally closed valves in each of the two lines between the HP pump suction header and the DH pump discharge lines are not acceptable as proposed. Figure 9.3-8 shows the valves as being manually operated. These valve should be motor operated valves with indication and control from the control room.

The last paragraph of the response to Request 6.27 states "the low pressure injection system is provided with flow-limiting devices in the cross-overs to restrict flow out the ruptured line, thus providing adequate flow through the intact core flood line." The PSAR does not contain information on either the cross-over lines or the restrictors. Provide drawings, analyses, plans for required testing, bases for instrumentation, and any other information pertinent to the specification of the cavitating venturis. Describe how the as-built flow split performance of the low pressure injection system will be evaluated during preoperational testing.

6.80 The response to Request 6.33 is inadequate. There have been ECCS and reactor design changes which are not accounted for in BAW-10048. It is our understanding that such changes are accounted for in BAW-10065 Supplement No. 1. If that topical report is applicable to Bellefonte, reference it, if not, provide Loss-of-Coolant accident analyses for Bellefonte.

15.0 ACCIDENT ANALYSES

15.15 The response to Request 15.1 is inadequate. Our evaluation of the list of engineered safety systems equipment required to operate in the MU & P system shows that the system is unsatisfactory from a single failure viewpoint for the following reasons (Refer to Figure 9.3-7): (1) the manually operated valves are normally closed on both the suction and discharge sides of high pressure pump P1C yielding an ineffective pump and precluding parallel suction from the BWST through valve HV47B, (2) the failure of valve HV47A to open results in the loss of all suction to the two HP pumps which are automatically aligned for high pressure injection, and (3) failure of valve HV48 to close results in air at the suction of the HP pumps after the makeup tank empties. Comment on this conclusion and change the design to accommodate these single failures.

Indicate which equipment is operating during normal plant operations. Extend the present list to include the ESF equipment required for long term heat removal and the switching from ECC injection to ECC recirculation. Indicate for each component in the complete list whether power is required and available from normal AC sources, emergency AC power, and DC batteries.

15.16 Indicate whether Figure 15.8-1 is the same as the two figures referred to but not attached to, Responses 15.6 and 15.7.

It is not obvious why curves such as Figure Q15.8-1 are applicable to the transients discussed in Responses 15.6, 15.7 and 15.9. The flow in Figure Q15.8-1 is assumed constant, yet flows are not constant during these transients. Keeping in mind the "parametric distortion" discussed in Section 6-4 of Boiling Heat Transfer and Two Phase Flow, L. S. Tong, John Wiley & Sons, New York, 1965, provide an estimate of the margin in hot channel flow parameters to demonstrate that a minimum DNBR of 1.32 is maintained for all transients discussed in Responses 15.6 through 15.9 for which the flow is not constant.